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P. A. Morris, Director, DRL

INDIAN POINT 2 REVIEW

A summary report of the ACRS Subcommittee meeting on Indian Point 2, held at O'Hare Airport on April 25, 1970, has been prepared by K. Kniel. A copy is enclosed for your information.

Original signed by R. C. DeYoung

R. C. DeYoung, A/D for PWR's
Division of Reactor Licensing

Enclosure:
Summary Report of Indian
Point 2 ACRS Subcommittee Mtg.

cc w/encl:
D. Muller, DRL
K. Kniel, DRL
M. McCoy, DRL
C. Long, DRL
K. Goller, DRL
R. Klecker, DRL
O. Parr, DRL
V. Moore, DRL

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DATE ▶	5/1/70					

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

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ACRS SUBCOMMITTEE MEETING

ON INDIAN POINT 2

I. Introduction

An ACRS Subcommittee meeting on Indian Point 2 was conducted on April 25, 1970, at Chicago O'Hare Airport. Attending for the staff were R. DeYoung, D. Muller, K. Kniel, M. McCoy, M. Rosen, H. Richings, V. Moore, and O. Parr. The ACRS was represented by D. Okrent, S. Hanauer, L. Squires and B. Kaufman. The ACRS Staff representative was M. Libarkin. W. Kipinski and E. Epler, ACRS consultants, were also present. The applicant was represented by 33 individuals headed by W. Cahill, R. DeYoung, M. Rosen, and H. Richings attended part-time departing about 4:00 p.m. The meeting terminated at 5:30 p.m.

II. Session with the Staff

Dr. Hanauer chaired the session. He said the major questions involved (1) the acceptability of the analyses of expected transients with failure to scram as submitted in Supplement 7 to the FSAR, and (2) the reliability of the protection system.

Item (1) was discussed and there appears to be several questions remaining to be answered. These include (the acceptability of the criterion for damage (DNB of 1.0 for the hot channel), (b) the ability to eventually terminate the transients, (c) the completeness of the transients selected, (d) the need to extend the loss of flow transient to include flow loss from the loss of more than one pump, including 2, 3, and 4 pumps, (e) the possible need to review the loss of feedwater pump(s) with partial loss of flow, and (f) the resolution of some apparent errors and discrepancies in the topical report.

Item (2) was discussed and directed to the question as to whether the Indian Point 2 protection system is acceptable and "up to snuff". V. Moore briefly summarized the issues and stated that O. Parr had a 30 minute presentation on the staff's review. Dr. Hanauer decided to keep item (1) and item (2) separate and have the applicant in on item (1) before the Subcommittee addressed item (2) and listened to O. Parr's presentation.

III. Session with the Staff and Applicant

The applicant made a presentation on the failure to scram following an anticipated transient study. J. Moore of Westinghouse made the presentation. He stated that in his opinion the analyses performed are more conservative than necessary and that Westinghouse was continuing its effort in the analyses mostly to avoid the continuing need for conservative assumptions. The analyses are useful to a designer, but the emphasis should be placed on the prevention of common-mode failures.

He also stated that the Westinghouse decision not to size the secondary steam system relief valve capability on the Section III permissive assuming reactor trips have proved to be a wise position. The consequences of several anticipated transients without reactor trips are tolerable due to the sizing of the valves. This raises the question for the staff as to how CE & B&W stand in this respect.

M. Mannigan then presented the details of some of the analyses. The initial discussion related to total loss of flow without scram. This analysis, is in addition to those given in the topical report, and one which the applicant does not intend to document. Following the initial transient the steady state conditions obtained are a power level of 52% of full power, a natural circulation flow of 10.6%. The minimum DNBR is 0.6, the peak clad temperature is 1350°F, and the steady state and clad temperature is about 1150°F. The steady state DNBR is greater than unity. The energy is dissipated by venting the primary system at the pressurized relief valve set pressure, and dissipation of the secondary system energy to the turbine which is assumed to trip only if the reactor trips. The analyses was extended only to 5 minutes.

Questions were raised and discussed concerning the fuel temperature, fuel rod failures, the mechanisms for loss of all pumps, mechanisms for the loss of flow without scram including gas turbine operation, and reactivity effects due to boron injection.

J. Moore summarized Westinghouse's conclusion that the loss of all 4 pumps without scram was tolerable for about 2-3 minutes but that a continued acceptable situation depended upon maintenance of a heat sink.

In connection with heat sink availability it was pointed out that normal steam generator feedwater inventory was equivalent to about 100 full power seconds of heat dissipation. This is always available regardless of the loss of power to the plant. If power is available to the condensate pumps then the condenser hot well inventory is also available as feedwater to the steam generators and this inventory is equivalent to 5 minutes at full power.

The Westinghouse position is that the Indian Point 2 plant design is acceptable with regard to anticipated transients because of the probability of scram but that additional analyses are being made and if minor design changes can be made to improve the overall situation then such changes will be considered for Indian Point 2.

Dr. Okrent suggested that it could be extremely advantageous if boron injection could be provided at high pressure instead of 1500 psi. It appeared from the discussion that it could be advantageous since the core would be shut down about the time the water inventory would need replenishment. Westinghouse said that this was being provided to some extent in new plants (Sequoyah and others) by manual actuation of the new centrifugal charging-injection pumps.

Westinghouse then described the analyses of a depressurization transient resulting from inadvertent opening one pressurizer safety valve. The transient was not presented in their topical report. The transient results in a reduction of primary system pressure to 1340 psi in 5 minutes and a steady state power level of 102%. The minimum DNBR is 1.5 to 1.2. There are no pump cavitation or heat sink availability problems. The analysis was terminated at 5 minutes. Dr. Hanauer queried Westinghouse as to subsequent consequences. Rods must be inserted, boron must be injected, or the open valve must be closed to prevent cavitation of the pumps and subsequent unacceptable consequences.

Dr. Hanauer began a review of the transients covered in the Westinghouse report. The loss of load turbine trip case was reviewed. The transient as calculated assumes that the four reactor coolant pumps are transferred from the Indian Point 2 generator power to the incoming 138 kV line. The probabilities for success of this transfer was discussed at length. It was concluded that there would be ample opportunity to verify this

procedure since a turbine trip (normally with reactor scram) could be expected in the average of at least twice a year. The results of this case are less severe than the loss of all pumps event. Steam generator feedwater availability (steam generator inventory plus condenser hot well inventory) is the limiting condition since secondary steam is ejected via safety valves. High pressure boron injection capability would mitigate the consequences of this case.

The next transient was excessive load increase. Dr. Hanauer conducted a general discussion of this event. The consequences were much less severe than the previous transients.

The next transient was loss of feedwater pump(s). The loss of a single feedwater pump is relatively trivial. The probability of losing both feedwater pumps was discussed. The consequences are severe. The lube oil system is common to both feedwater pumps. The cooling system has common components. The SG level control system failure was described. J. Moore said this could be considered to be an anticipated transient. It has not been reviewed for Indian Point 2.

The next transient was loss of one primary coolant pump. This is an acceptable transient. There were no questions on this since the more severe case of loss of all pumps previously discussed is a more limiting case.

The next transient was the rod withdrawal event. Westinghouse discussed the basis for selecting the full power initial condition. The selection appeared to be reasonably conservative even though an initial low power condition would result in more overall reactivity insertion and a higher power and temperature. The higher temperatures of the fuel rod could lead to fuel failure mechanisms involving fuel melting and releases of gap activity. Westinghouse is continuing work on this.

The next transient was the startup event. Westinghouse chose to make the calculation for a relatively low worth bank but fast acting as opposed to a high worth slow-acting bank as this was considered to be more conservative. The results were less severe than those for other events.

The next transient was the station blackout. The results of this are similar to the severe results of the loss of all primary pumps.

The next transient was the startup of an inactive loop. The consequences are not severe.

IV. Session with Staff

The Subcommittee met with the staff again to discuss the second major item on the Subcommittee's list of considerations.

Olan Parr made a rather detailed presentation of the poorer design aspects related to the IP-2 protection and electrical systems. This included discussion of the single tunnel, the ESF manual actuation panel in the control room without separation in the panel, the common diesel location in a non-tornado proof structure, the sensing of undervoltage to start the diesels, cable separation, cable penetrations at the containment, and automatic trip point change requirements for operation on three loops.

The Subcommittee was "appalled" at the situation. They asked if we did not have an Oyster Creek situation in hand and whether we should not have the applicant make an independent review of his work as we required of Jersey Central. The staff stated that an Oyster Creek situation did not exist since we felt we knew of all significant problems on this plant. The Subcommittee asked the staff's opinion about the advisability of a Subcommittee review at the site. We stated that a review of that type could be extremely useful.

During the "break" prior to inviting the applicant to return, Dr. Okrent informed me that:

(1) He would be in Europe from 6/15/70 to 8/9/70. He is thinking about flying back for the July meeting if Indian Point 2 is on the agenda. His return is in time for the August meeting.

(2) He suggested that the staff report review all items it was not completely content with, describe the changes that could be made to satisfy us and indicate the ease or difficulty of the change.

(3) He discussed a "high capacity-high pressure" safety injection system requirement for PWR plants and suggested that an "opening" be left in our position on plants such as Midland for such a requirement.

V. Session with Staff and Applicant

Dr. Hanauer led a lengthy question and answer session on possible mechanisms for achieving a failure to scram event.

Dr. Okrent and Hanauer asked Westinghouse to provide information on potential weaknesses in the reliability of the scram system. In their replies they indicated their awareness of the importance of reliable design for the single scram bus system as used in all their current generation plants. They were unwilling to admit to any points of weakness in this design. This answer did not satisfy the Subcommittee.

There was considerable discussion as to what the operator could do to affect scram in the several minutes available to him for the case of severe transients which ended in a total loss of heat removal capability. The operator could go to the location of the scram trip breaker and actuate it manually. In most plants this could be done in less than two minutes. The power breakers supplying each of the two motor generator sets providing control rod power at 57 Hertz could also be tripped from either the control room or locally.

Dr. Hanauer asked if power could potentially be available from any other source to hang up the rods after a trip. The answer was negative, but he persisted in questioning the details regarding wiring and control and power for the control rod drive control package. As this question was not on the original agenda the appropriate Westinghouse people were not present and no answer was forthcoming. J. Moore referred the Subcommittee to the WCAP report covering the Westinghouse rod control system.

There was further discussion concerning the availability of scram circuits for tests. These circuits are available for test by opening panels in the area adjacent to the control panel. These areas are normally not available to direct surveillance by the operator on duty. After much previous discussion the applicant had agreed to provide separate annunciator lights in the control room for each panel which would light up when the panel was opened for test or any other purpose. Administrative control by the operator is thus required to assure that only a limited number of circuits are out for service or test at any given moment. Although this subject has been discussed repeatedly on many different occasions no concept that would not involve administrative control has been suggested by anyone.

Dr. Okrent asked about scram diversity for the small break loss-of-coolant accident. Analysis indicates that scram is required to shut down the reactor for break sizes below 4.5 ft^2. Above that size break to the doubled ended break ($\sim$9 ft^2) voiding can be relied on to shut the reactor down. For break sizes equivalent to less than a 4 inch diameter line the thermal load can also be handled without scram. Presumably the cooling provided by the emergency core cooling system in this range is adequate until power is reduced by the borated ECCS injection. Between the break sizes of 4.5 ft^2 and that equivalent to a 4 inch diameter line the diverse signal for reactor trip (i.e., other than low pressurizer pressure) is high containment pressure. Westinghouse has not performed the core thermal transient analysis for this size range using the high containment pressure signal for reactor trip.